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Team Leader:	R. Gibbs, Senior Reactor Inspector Maintenance Branch Division of Reactor Safety, RII
Inspectors:	J. Coley, Reactor Inspector, Maintenance Branch, RII M. Franovich, Resident Inspector, McGuire, RII M. Miller, Reactor Inspector, Engineering Branch, RII F. Talbot, Operations Engineer, NRR S. Wong, NRC Contractor

Approved By: P. T. Frederickson 10/3/97

P. Fredrickson, Chief
Maintenance Branch
Division of Reactor Safety

Date Signed

EXECUTIVE SUMMARY

McGuire Nuclear Plant, Units 1 and 2
NRC Inspection Reports 50-369/97-14 and 50-370/97-14

This inspection included a review of the licensee's implementation of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" [the Maintenance Rule]. The report covers a one-week period of inspection by inspectors from Region II.

Overall, the inspection team concluded that the licensee had a comprehensive Maintenance Rule program, and the program was being effectively implemented. The team found only minor deficiencies in program implementation, which were immediately corrected by the licensee. No violations or other open items were identified by the inspection.

Operations

- Personnel interviewed (licensed operators, work control center personnel, an STA, and a scheduler), in general, understood their specific duties and responsibilities for implementing the Maintenance Rule (Section O4.1).

- Personnel interviewed (licensed operators, work control center personnel, an STA, and a scheduler) understood the risk-matrix and its limitations for removal of equipment from service (Section O4.1 and M1.2).

Maintenance

- Required structures, systems, and components (SSCs) were included within the scope of the Rule with the exception of the non-safety related equipment needed to assure a reliable source of water and provide protection for the AFW pumps (Section M1.1).

- Two examples of scoping enhancements concerning lower containment ventilation and the ice condenser systems were observed during the inspection, which demonstrated a proactive approach toward adjustment of the Maintenance Rule program-based on site specific experience (Section M1.1).

- Plans for performing periodic evaluations met the requirements of the Rule (Section M1.3).

- A weakness was identified concerning trending of system data, in that, no uniform method existed to track and report system status, and status reports did not exist for all Maintenance Rule systems (Section M1.6 and M1.7).

- The approach to balancing reliability and unavailability was reasonable (Section M1.4).

- The licensee had considered safety in establishing goals and monitoring for systems and components in (a)(1) status (Section M1.6).

- Review of SSCs in (a)(2) status determined that performance criteria were established commensurate with safety (Section M1.7).

- Industry-wide operating experience was used (Section M1.6 and M1.7).

- The structures program established under the Rule was assessed as a strength because it was comprehensive and was being effectively implemented (Section M1.7).

- In general, walkdown of systems determined that the systems were being satisfactorily maintained (Section M2.1).

- Audits and self-assessments of the Maintenance Rule program were thorough and corrective actions were appropriately implemented. Audits and assessments were considered to be a strength (Section M7.1).

Engineering

- The overall approach to performing risk-ranking for SSCs within the scope of the Maintenance Rule using the probabilistic risk-assessment (PRA) and the expert panel was satisfactory (Section M1.2).

- The PRA truncation limit was not low enough to ensure risk-significant SSCs were not omitted. This issue was identified and was being addressed by the licensee prior to the inspection (Section M1.2).

- The expert panel's decision making process was based on consensus judgement which was considered sufficient (Section M1.2).

- Use of system engineers to provide technical details during the meetings was assessed as a strength (Section M1.2).

- The risk-matrix and associated procedure for removal of equipment from service were satisfactory (Section O4.1 and M1.5).

- The risk-matrix was developed by analyzing two systems out of service at one time, and had not been analyzed for a greater amount of systems taken out of service (Section O4.1 and M1.5).

- Systems engineers' knowledge of their systems was excellent and their knowledge of the Maintenance Rule was adequate. Knowledge of Maintenance Rule responsibilities under Duke Power Procedure EDM 210 was was not strong (Section E4.1).

Report Details

Summary of Plant Status

Units 1 and 2 operated at power during the inspection period.

Introduction

The primary focus of this inspection was to verify that the licensee had implemented a maintenance monitoring program which met the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (the Maintenance Rule). The inspection was performed by a team of inspectors that included a team leader and three Region II based inspectors, an operations engineer from NRR, and an NRC contractor. The licensee provided an overview presentation of the program to the team on the first day of the inspection. The overview handout is included as an attachment to this report.

I. OPERATIONS

O4 Operator Knowledge and Performance

O4.1 Operator Knowledge of Maintenance Rule

a. Inspection Scope (62706)

During the inspection, the team interviewed one licensed senior reactor operator (SRO), a shift technical advisor and an Operations Shift Manager who were also SROs, two individuals from the Work Control Center, and one scheduler to determine if each understood the general requirements of the Maintenance Rule and the particular duties and responsibilities of their respective positions for its implementation.

b. Observations and Findings

The operators were responsible for the following tasks associated with the Maintenance Rule:

- determining the impact on availability of SSCs when tagging equipment out-of-service and performing administrative requirements for tagging;
- determining SSC out-of-service logging requirements and impact on availability;
- evaluating priorities for system restoration;
- evaluating job scheduling activities, and
- evaluating plant configuration to determine if work authorization created undue risk.

In general, the operators interviewed understood the philosophy of the Maintenance Rule and their responsibilities associated with the Rule. The operators all believed that they were adequately trained and understood the requirements of the applicable procedures. All operators understood the need to restore equipment to operating condition and minimize SSC unavailabilities.

The team reviewed control room logs for an operating period of two months for each of the two plant units. The log entries provided sufficient documentation to extract information on out-of-service times for inoperable plant equipment. The documentation details also allowed the log entries to be easily used as a source of information for Maintenance Rule recordkeeping.

The PRA matrix was a tool used by operators, work week managers, and schedulers to assess risk when removing equipment from service. The team interviewed operators, work control center SROs and schedulers to determine their familiarity with the use of the matrix and the level of knowledge regarding the limitations of the matrix. The PRA matrix was used by tne SROs on shift, the work week managers, and work control staff. Those persons interviewed understood the use of the PRA Matrix as defined by WPM-607, "Maintenance Rule Assessment of Equipment Removed from Service," Revision 3. As a result of a weakness identified by earlier NRC inspections at the Catawba and Oconee nuclear plant sites, a Duke Power Company memorandum on limitations of the PRA matrix had been provided to the operators, work control center SROs, and schedulers. The personnel interviewed were aware that the matrix might not provide an accurate assessment of risk when more than two out-of-service functions were affected. Also, these personnel were aware that removing items from service that were not on the matrix could impact plant-risk. Currently, the licensee's staff is planning for the implementation of personal computer-based PRA software (i.e., the SENTINEL program) for on-line risk-assessment of concurrent equipment outages. The use of this tool should enhance the process for proactive evaluations of the risk-impact of multiple SSCs being out-of-service within the same time period.

c. <u>Conclusions</u>

In general, the SROs, the Operations shift manager, work control center SROs, work week managers, and schedulers interviewed clearly understood the philosophy of the Maintenance Rule and their specific responsibilities for implementation of the Rule. These personnel were familiar with the Maintenance Rule procedures and the PRA Matrix. Also, they were aware that the PRA matrix did not provide adequate information on risk-impact of plant configurations with more than two out-of-service SSCs.

II. MAINTENANCE

M1 Conduct of Maintenance

M1.1 Scope of Structures, Systems, and Components Included Within the Rule

a. Inspection Scope (62706)

Prior to the onsite inspection, the team reviewed the McGuire Updated Final Safety Analysis Report (UFSAR), Licensee Event Reports (LERs), the Emergency Operating Procedure (EOPs), previous NRC Inspection Reports, and other information provided by the licensee. The team selected an independent sample of SSCs that the team believed should be included within the scope of the Rule, which was not classified as such by the licensee. During the onsite portion of the inspection, the team used this independent sample to determine if the licensee had adequately identified the SSCs that should be included in the scope of the Rule in accordance with 10 CFR 50.65(b).

The team's review was performed to assure the scoping process included:

- all safety-related SSCs that were relied upon to remain functional during and following design basis events to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10 CFR, Part 100 guidelines;
- non-safety related SSCs that were relied upon to mitigate accidents or transients;
- non-safety related SSCs which were used in the plant EOPs;
- non-safety related SSCs whose failure could prevent safety-related SSCs from fulfilling their function, and;
- non-safety related SSCs whose failure could cause a reactor trip or actuation of a safety-related system.

b. Observations and Findings

The licensee reviewed a total list of 247 systems and 93 structures at the McGuire Nuclear Station to determine which SSCs were within the scope of the Maintenance Rule. The licensee determined that 170 systems and 48 structures were within the scope of the Maintenance Rule. In addition, the licensee reviewed 562 functions and determined that approximately 365 functions were in the scope of the Rule. The team reviewed the licensee's database of SSC functions excluded from the scope of the Maintenance Rule and verified that appropriate technical justification existed to exclude these SSCs.

The team determined that all required SSCs were included in the Rule with the exception of the following.

- The team determined that the licensee had not included non-safety related AFW equipment used in the EOPs to mitigate the consequences of an accident and provide protection for the AFW pumps in the scope of the Maintenance Rule. Specifically, the licensee did not identify that operators would, per the EOPs, isolate the non-safety AFW water sources upon each tank's depletion. EOPs that rely upon the AFW system require operators to isolate the AFW condensate storage tanks upon low level and to isolate the upper surge tanks upon low level to prevent airbinding (a common mode failure) of the AFW pumps.

  In response to this finding, the licensee reviewed the issue during an expert panel meeting held during the inspection. A recent hydraulic study of the non-safety water systems interaction with the safety portions of AFW indicated that isolation of some of the non-safety water tanks might not be required. However, the panel rescoped the AFW system to include the non-safety related equipment and concluded that the non-safety systems could fail a safety related system. The team considered the licensee's actions demonstrated strong initiative to address the issue and proceeded accordingly with corrective actions.

- The licensee had not included a shutdown function for the refueling water system regarding accident mitigation upon a loss of Residual Heat Removal (RHR) cooling. Specifically, Abnormal Operating Procedure "Loss of RHR or RHR System Leakage" AP/2/A/5500/19 instructs operators to makeup Reactor Coolant System (RCS) inventory by gravity feed from the refueling water systems through the opening of valve 2FW27A. The team considered this issue a weakness and encouraged the licensee to examine more closely the "important to safety" shutdown functions of plant SSCs used for accident mitigation. The licensee responded that they would examine the issue and also readdress the SSCs summary functions sheets when the shutdown PRA was completed. Later, the licensee also informed the team that the SSC function sheet referenced a selected licensee commitment for gravity feeding capability from the refueling water storage tank. However, the licensee indicated that they would examine if clarifying language should be added to the refueling water system functions.

The team noted two examples of scoping enhancements which were identified by the licensee prior to the inspection and were corrected during the inspection.

- The lower containment ventilation system (LCV) was scoped into the Maintenance Rule program. The system engineer and the licensee's recent self-assessment identified that this system should be evaluated for inclusion in the McGuire Maintenance Rule program. According to the UFSAR, the LCV did provide some minor accident mitigation functions for very small break loss of coolant accidents (LOCAs), mostly below the threshold of ice condenser system actuation. The expert panel convened on August 20, 1997, and concluded that the LCV system should be within the Maintenance Rule program because of the statements contained in the UFSAR.

- The ice condenser system function NF.05 was clarified to include the floor cooling system. The system engineer requested that the Maintenance Rule coordinator and expert panel clarify function NF.05 (air handling units) to specify the floor cooling system in view of a recent event where 10 of 48 lower inlet doors were inoperable. The system engineer considered the floor cooling system as a subsystem of the air handling units. This event was caused by the concrete floor heaving upward to the point where door frame flashing caused the affected doors to become mechanically bound. The licensee attributed the problem to previous operational events where ice melted and the water was absorbed into the concrete. During the last operating cycle, the floor cooling system became degraded and may have aggravated this condition (see Section M1.7 for details). The panel approved this addition to the NF.05 function in the expert panel meeting held on August 20, 1997.

c. Conclusions

In general, required SSCs were included within the scope of the Rule. A deficiency was identified for failure to include the non-safety related equipment relied upon in the EOPs to assure a reliable source of water and to provide protection for the AFW pumps. A weakness was noted concerning the lack of a shutdown function for the refueling water system regarding accident mitigation upon a loss of RHR cooling.

Two examples of scoping enhancements concerning the LCV system and the ice condenser were observed during the inspection. These enhancements demonstrated a proactive approach toward adjustment of the McGuire Maintenance Rule program based on site specific experience.

M1.2 Safety or Risk Determination

a. Inspection Scope (62706)

Paragraph (a)(1) of the Maintenance Rule requires that goals be commensurate with safety. Implementation of the Rule using the guidance contained in NUMARC 93-01 also requires that safety be taken into account when setting performance criteria and monitoring under (a)(2) of the Rule. This safety consideration is used to determine if the SSCs should be monitored at the train or plant level. The team reviewed the methods that the licensee had established for making these required safety determinations. The team also reviewed the safety determinations that were made for the systems that were reviewed in detail during this inspection.

b. Observations and Findings

The licensee's expert panel was established in accordance with Section 9.3.1 of NUMARC 93-01. The expert panel determined which functions were within the scope of the Rule, evaluated the risk-significance ranking of SSCs, and established the performance criteria for SSCs. Members of the expert panel included representatives from operations, maintenance, engineering (electrical, mechanical, and civil/structural engineers), and the PRA group. The total experience of the expert panel members was over 110 man-years of nuclear industry experience. In addition, the panel members were either registered professional engineers or were previously licensed SROs. All panel members were from the station staff except for the PRA representative who was from the corporate office. The team noted that the expert panel members who did not have a strong PRA background had received some PRA training. The team also observed that the conduct of the expert panel meeting complied with administrative procedure guidance.

The expert panel determined the risk-significance of SSC functions based on the combined results from PRA and deterministic considerations. The expert pane, s decision-making process was based on consensus judgment. The McGuire PRA provided information on PRA importance measures used for risk-ranking of SSCs. The importance of measures used were risk achievement worth, risk reduction worth, and cutsets contribution to 90% of core damage frequency which were consistent with the guidance provided in NUMARC 93-01. To allow for discussion of all cutsets, high risk cutsets containing human actions were not screened out prior to evaluation by the panel. The expert panel removed 13 SSC functions from the list of SSCs which had met at least one of the quantitative criteria for risk-significance. The team reviewed this list and concurred with the expert panel's reasoning on 11 SSC functions. The licensee was continuing further review on the risk-significance of two SSC functions related to control of the plant tie breakers in the 525 KV Switchyard and 230 KV Switchyard. At the time of the inspection, the expert panel had declared 51 SSCs to be risk-significant. Except for the above-mentioned switchyard tie breakers, the team did not identify any SSCs that had been improperly ranked.

b.1 Risk-ranking

The team reviewed a sample of SSCs within the scope of the Rule that the expert panel had categorized as non-risk significant to assess if the expert panel had properly determined the safety significance of those SSCs. In general, the team found that the expert panel had properly categorized the safety significance of SSCs and had documented the basis of their conclusions. The team determined that the licensee had considered initiating events and selected recovery actions in the ranking process. Also, the licensee's expert panel had evaluated the risk-significance of containment SSCs, and containment isolation valves and hydrogen igniters were categorized as risk-significant SSCs.

The information used for risk-ranking SSCs was based on the PRA model developed to support the 1991 Individual Plant Examination, and 1994 Individual Plant Examination for External Events studies submitted to the NRC. Generic failure data and plant-specific data for component failures and unavailabilities from 1985 to 1990 were used in the PRA calculations. The PRA database of basic event failure rates and unavailabilities had not been updated during initial implementation of the Maintenance Rule. At the time of this inspection, the licensee's PRA staff was in the process of preparing Revision 2 of the PRA which was scheduled for completion in December 1997. The revised PRA will contain updated SSC failure information based on data collected from 1989 to 1995. The cutsets generated from the original PRA model were the basis for Maintenance Rule evaluations. The PRA model was a "pre-solved cutsets" model containing cutsets of accident sequences initiated by internal events (e.g., loss of offsite power) including fires, floods and tornadoes. Seismic event-initiated sequences were not included to prevent "shadowing" of important events. The licensee was in the process of implementing PC-based software to perform full requantification of the PRA model for Maintenance Rule evaluations.

A truncation level of 1E-8 was used to quantify the PRA results used for risk-ranking. This truncation limit was about two and one-half orders of magnitude less than the overall core damage frequency estimate of 6E-5/reactor-year core damage frequency. This limit was higher than that which was normally considered desirable to ensure that risk-significant SSCs were not omitted from risk-ranking considerations. As a result of similar findings from NRC inspections at the Catawba and Oconee nuclear plant sites, the licensee had initiated a Problem Investigation Process report (PIP) 0-G97-0124 to address this issue. At the time of the inspection, the licensee was performing sensitivity studies to determine whether using a cutset-frequency truncation value of 1E-8 would include all risk-significant SSCs. The licensee planned to use lower truncation levels when the PC-based software for on-line risk-assessment is implemented in the near future.

b.2 Performance Criteria

The team reviewed the performance criteria to determine if the licensee had adequately established performance criteria under (a)(2) of the Maintenance Rule. Section 9.3.2 of NUMARC 93-01 recommends that performance criteria for risk-significant SSCs be set to assure that the availability and reliability assumptions used in the risk-determining analysis (i.e., PRA) were maintained.

The team reviewed Calculation MCC-1535.00-00-0008, "PRA Analysis of Maintenance Rule Availability Performance Criteria," approved May 21, 1996. This document presented the licensee's methodology for establishing availability limits based on risk-significance of the SSC functions. Availability limits were set at 98%, 96%, and 94% availability for high, medium, and low risk-significant categories of SSCs on the basis of risk-achievement worth values. In the case of SSC functions without analyzed risk-achievement worth values, the expert panel would review the risk-significance category of the SSC function to determine the appropriate performance criteria. The team noted that PRA sensitivity analyses were performed to evaluate the cumulative risk-impact of setting the SSC unavailabilities equal to the proposed unavailability limits for all risk-significant SSCs. The sensitivity analysis results showed a small increase in core damage frequency when the unavailabilities for all of the risk-significant SSCs were assumed to be simultaneously at half of their allowable values. The use of half of the unavailability criteria was based on a reasonable assumption that 50% of SSC unavailability was for planned maintenance, and 50% of unavailability was for unscheduled downtime resulting from random failures. The team noted that the unavailability criteria for some risk-significant SSCs (e.g., turbine-driven AFW pump, RHR pump train, and safety injection (SI) pump train) were less stringent than the unavailability assumptions used in the PRA. During the inspection, the team requested that the licensee staff perform additional sensitivity analyses to evaluate the cumulative risk-impact of setting the unavailabilities of the selected SSCs simultaneously equal to their proposed performance criteria limits. The calculated core damage frequency increase was about 7% above the plant baseline. Given that the estimated plant core damage frequency value was 6.0E-5/reactor year, this increase was within the Electric Power Research Institute (EPRI's) Probabilistic Safety Assessment (PSA) Application Guidelines for core damage frequency incremental limit (i.e., about 13%) allowed for a risk-significant change. Based on insights from the sensitivity analyses, the licensee agreed to consider revising the unavailability criteria for the turbine-driven AFW pump and SI pump train to more stringent values.

The licensee elected to use reliability performance criteria that counted maintenance preventable functional failures (MPFFs) at the system and function levels. The acceptable limit on MPFFs varies from zero to five MPFFs per operating cycle (about 18 months) depending on the availability limits in consonance with the risk-significance of the SSC functions. A limit of three MPFFs was used as reliability performance criteria for all risk-significant SSCs, and a limit of five MPFFs was used for the non-risk significant SSCs. For the SSC functions reviewed, the team found that restrictive availability criteria for some risk-significant SSCs would compensate for the less restrictive reliability criteria. Specifically, in cases of those SSC functions where one MPFF per operating cycle was not acceptable, the licensee had set the required availability to 100%. The team noted that the licensee had not yet performed a sensitivity analysis to demonstrate that the proposed reliability performance criteria would not have a significant impact on total core damage frequency. However, reliability curves for some failure rates (e.g., 1E-3 or 1E-2) assumed for SSC reliability had been developed for use to determine MPFF limits. The licensee's approach for establishing the reliability performance criteria for standby SSCs was based on a reasonable estimate of SSC demands during the monitoring interval. The team considered that the licensee's approach to setting performance criteria was acceptable.

b.3 Expert Panel

The team reviewed the licensee's process and procedures for establishment of an expert panel. It was determined that the licensee had established an expert panel in accordance with the guidance provided in NUMARC 93-01. The expert panel's responsibilities included the final authority for decisions regarding Maintenance Rule scope, risk-significance, and performance criteria selection.

Duke Power Procedure EDM-210, "Requirements for Monitoring the Effectiveness of Maintenance of Nuclear Power Plants or the Maintenance Rule," Revision 5, contained the guidance regarding expert panel activities, member qualifications and conduct of the expert panel meetings. The panel was comprised of personnel from the site as well as personnel from the Duke Power Company General Office.

The team observed an expert panel meeting held during the inspection (on August 20, 1997). The team noted that the use of system engineers to provide technical details during expert panel meetings was a strength. Panel members were questioned on previous decisions and aspects of panel responsibilities. The expert panel members interviewed had an adequate working knowledge of their responsibilities with respect to Maintenance Rule implementation.

c. Conclusions

The licensee's overall approach to performing risk-ranking of SSCs for the Maintenance Rule was satisfactory. The licensee's performance criteria for reliability and unavailability of SSCs was commensurate with assumptions in the PRA. The expert panel's decision-making process was based on consensus judgment which was considered sufficient. The use of system engineers to provide technical details during the expert panel meetings was assessed as a strength.

The PRA truncation limit was not low enough to ensure that risk-significant SSCs were not omitted for risk-ranking purposes. However, this issue was identified and was being addressed by the licensee staff prior to the inspection.

M1.3 Periodic Evaluation

a. Inspection Scope (62706)

Paragraph (a)(3) of the Rule requires that performance and condition monitoring activities and associated goals and preventive maintenance activities be evaluated taking into account, where practical, industry-wide operating experience. This evaluation is required to be performed at least one time during each refueling cycle, not to exceed 24 months between evaluations. The team reviewed the procedure the licensee had established to ensure this evaluation would be completed as required. In addition, the team discussed the requirements with the Maintenance Rule coordinator, who was responsible for this activity.

b. Observations and Findings

At the time of inspection, the license had not completed the first periodic assessment. Cycle 11 for Unit 1 was completed on May 14, 1997. This was McGuire's first refueling outage since implementation of the Maintenance Rule. A periodic assessment as required by 10 CFR 50.65 (a)(3) was initiated for Unit 1. However, a Duke Power self-assessment team reviewed the incomplete draft in July 1997, and concluded that several considerations specified in Procedure No. EDM-210 and NUMARC 93-01 were not addressed. In addition, the team was critical of the incomplete draft in that, it presented a significant amount of detail information without drawing meaningful conclusions in many areas regarding the program's effectiveness. The self-assessment team recommended that the guidance in NUMARC 93-01, Revision 2, be followed for conducting the evaluation. This guidance states that multi-unit sites may perform the periodic assessment for both units concurrently, at an interval approximately equal to the length of the refueling cycle. This recommendation was incorporated, and the licensee indicated an intention to perform the first periodic assessment during the first quarter of 1998, after Unit 2 completed its Cycle 11 outage on December 25, 1997.

c. Conclusions

Based on discussions with the Maintenance Rule coordinator regarding the status of the periodic assessment, and review of Procedure EDM-210 for periodic assessment the team concluded that licensee plans for performing periodic evaluations met the requirements of the Rule.

M1.4 Balancing Reliability and Unavailability

a. Inspection Scope (62706)

Paragraph (a)(3) of the Rule requires that adjustments be made where necessary to assure that the objective of preventing failures through the performance of preventive maintenance is appropriately balanced against the objective of minimizing unavailability due to monitoring or preventive maintenance. The team met with the Maintenance Rule coordinator, system engineers, and representatives of the expert panel to discuss the licensee's methodology for balancing reliability and unavailability.

b. Observations and Findings

The team reviewed the licensee's approach to balancing system reliability and unavailability for risk-significant SSCs to achieve an optimum condition. The licensee had scheduled balancing reviews during periodic evaluations at refueling outages, not to exceed 24 months. The requirements for balancing reliability and unavailability were discussed in EDM-210, "Requirements for Monitoring the Effectiveness of Maintenance of Nuclear Power Plants or the Maintenance Rule," Revision 5. The Maintenance Rule coordinator was responsible for the balancing process for risk-significant SSCs during periodic system evaluations. The Maintenance Rule coordinator was also responsible for collecting data from the system engineers, who monitor and trend the system performance continuously.

The licensee's approach to balancing equipment reliability and unavailability consisted of establishing goals and performance criteria for the appropriate SSCs and functions and then monitoring the performance of the affected equipment. An implicit assumption was made that, if appropriate goals and criteria were set and if such goals and criteria were met, then an appropriate balance between unavailability and reliability would be achieved.

The team concluded that such an approach should provide a reasonable balance, provided that appropriate goals and performance criteria were always established.

The licensee had conducted an assessment to determine the impact of McGuire specific experience on the calculated core damage frequency. The team reviewed the assessment report, "PSA Assessment of MNS Unit 1 Cycle 11 Maintenance Rule

Experience," Revision 1, dated August 15, 1997. The evaluation of SSC failures was based on data from McGuire Unit 1 operating Cycle 11 for unavailability, functional failures, human errors and plant transients. The team noted that the reliability calculations included estimated SSC demands when actual data was not available. The results of the analysis indicated a calculated core damage frequency of 4.8E-5/reactor year. This represents about 20% reduction in the core damage risk for the operating cycle when compared to the McGuire baseline core damage frequency estimate of 6.0E-5/reactor year. Based on the results of this assessment, the licensee determined that reliability and availability of risk-significant SSCs were adequately balanced, and that no adjustments were necessary to the Maintenance Rule performance criteria. The team concurred with the licensee's assessment.

c. Conclusions

The team concluded that the licensee's method of balancing reliability and unavailability provided a reasonable approach to meet the intent of Section (a)(3) of the Rule.

M1.5 Plant Safety Assessments Before Taking Equipment Out of Service

a. Inspection Scope (62706)

Paragraph (a)(3) of the Maintenance Rule states that the total impact on plant safety should be taken into account before taking equipment out of service for monitoring or preventive maintenance. The team reviewed the licensee's procedures and discussed the process with the PRA representative, plant operators, an Operations shift manager, schedulers, and work control center SROs.

b. Observations and Findings

The team reviewed the licensee's process for removing equipment from service. The process was documented in Procedure WPM-607, "Maintenance Rule Assessment of Equipment Removed from Service," Revision 3, for removing equipment from service while the plant is at full-power operation. When the plant was shut down for refueling outage, procedural guidance for removing equipment from service was provided in McGuire Site Directive 403, "Shutdown Risk Management Guidelines," Revision 7.

When the plant was at full-power operation, the "McGuire PRA Matrix" (Attachment 607.6.5 in WPM 607) was used by schedulers and work week managers to evaluate plant-risk for single and double equipment outages. A 12-week rolling schedule was used for planning surveillance and preventive maintenance of plant equipment. The schedulers stated that the PRA matrix was used to prevent planned concurrent equipment outages which would place the plant in a high risk situation or in conditions which would violate plant Technical Specifications (TS). The work week managers and SROs stated that the PRA matrix was used for evaluating emergent work situations (resulting from unanticipated equipment failures). For combinations of equipment outages not considered in the PRA matrix, all personnel interviewed stated they relied on experience and judgment to evaluate plant-risk. As a result of weaknesses identified by earlier NRC inspections at the Catawba and Oconee nuclear plant sites, the licensee had issued a memorandum on limitations of the PRA matrix to operators, work control center SROs, and schedulers. All users of the PRA matrix interviewed were aware that the matrix did not provide an accurate assessment of risk when combinations of greater than two systems are out-of-service within the same time window.

The PRA matrix was used to assess risk-impact of component or train outages by considering the removal of system functions. As such, the matrix did not provide accurate guidance for risk evaluations of component or train level outages. Thus, the matrix was considered to be less than effective use of PRA information for evaluating plant-risk due to concurrent equipment outages. The specific issues concerning the PRA matrix were as follows.

- The matrix was constructed by the expert panel using only a qualitative assessment of risk for SSC functions taken out two at a time. At the time of inspection, the licensee staff provided quantitative assessment results for the various combinations identified on the matrix.

- Neither the matrix nor WPM-607 provided guidance for assessing true plant-risk when three or more matrix functions were affected at the same time. Combinations of multiple, low risk-significant SSCs removed from service might place the plant in a risk-significant configuration. There were no procedural restrictions on the number of SSC functions or SSCs that could be removed from service concurrently.

The licensee had initiated PIP 0-G97-0124 to address the above issues concerning the use of the PRA matrix. At the present time, the licensee staff was planning for the implementation of PC-based software (i.e., the SENTINEL code) for on-line risk-assessment of concurrent equipment outages. The implementation of this tool should enhance the proactive evaluations of risk-impact of multiple SSCs being out-of-service at the same time.

Shutdown risk was managed through McGuire Site Directive MSD-403, "Shutdown Risk Management Guidelines," Revision 7. Appendix D of this directive provides a discussion of the "Unit Shutdown Configuration Control Matrix" which addressed guidance for equipment-configuration control during a plant shutdown evolution. Risk was managed through the use of a key safety functions process that was defined in NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management." The licensee PRA group was planning for the implementation of a PC-based risk-assessment program which will include the outage risk-assessment manager models for assessing the risk of system unavailability during plant shutdown conditions.

As noted previously, 10 CFR 50.65 (a)(3) requires an assessment of the total plant equipment out-of-service to determine the overall effect on performance of safety functions during the performance of monitoring and preventive maintenance activities. The team reviewed the Technical Specifications (TS) Action Item Logs and control room operators logs over a two-month period for McGuire Unit 1 to determine risk-significant "time windows" in which several SSCs were concurrently out of service. The review period was from May 1 through June 30, 1997. The team identified three instances (May 22, June 5, and June 17, 1997) where configurations of more than four SSCs were out-of-service. These equipment-outage configurations were the result of planned maintenance and surveillance activities, and the work control staff had used the PRA matrix in scheduling the activities. The licensee was requested to evaluate the risk-impact of the three equipment-outage configurations in terms of core damage frequency and core damage probability estimates. The results of the risk evaluations indicated that there were no unacceptable risk due to the changed configurations during the sampled time period. Core damage probability estimates of the configurations were less than 1E-6, which was the risk-impact threshold defined in EPRI's PSA Applications Guidelines). However, the implementation of the PC-based SENTINEL software would enhance the licensee's process for on-line risk-assessment of equipment-outage configurations on a daily basis.

c. Conclusion

The licensee's procedure for removing equipment from service for maintenance was satisfactory. The users of the PRA matrix were aware of the limitations of the matrix for risk-assessment of equipment-outage configurations. The licensee staff was planning for the implementation of PC-based software for on-line risk-assessment of equipment-outage configurations to address the issues concerning the use of the PRA matrix.

M1.6 Goal Setting and Monitoring for (a)(1) SSCs

a. Inspection Scope (62706)

Paragraph (a)(1) of the Rule requires, in part, that licensees shall monitor the performance or condition of SSCs against licensee established goals, in a manner sufficient to provide reasonable assurance the SSCs are capable of fulfilling their intended functions. The Rule further requires goals to be established commensurate with safety and industry-wide operating experience be taken into account, where practical. Also, when the performance or condition of the SSC does not meet established goals, appropriate corrective action shall be taken.

The team reviewed the systems and components listed below for which the licensee had established goals for monitoring of performance to provide reasonable assurance the system or components were capable of fulfilling their intended function. The team evaluated the use of industry-wide operating experience, monitoring of SSCs against goals, and corrective action taken when SSCs failed to meet goal(s), or when as SSC experienced an MPFF.

The team reviewed program documents and records for the three systems or components the licensee had placed in the (a)(1) category in order to evaluate this area. The team also discussed the program with the Maintenance Rule coordinator, system engineers, and other licensee personnel.

b. Observations and Findings

b.1 Refueling Water System

The refueling water system on Unit 2 had been classified as (a)(1) on August 31, 1996, because it failed to meet its Cycle 10 availability performance criteria. Investigation by the licensee determined that the excessive unavailability was primarily caused by problems relating to inadequate heat tracing, resulting in freezing of system components affecting the swap over suction source for the unit's safety injection systems (reference PIP 2M96-0332 and LER No. 370-96-01). The team reviewed the corrective actions, goals and monitoring for the system and concluded that they were being appropriately implemented. In addition, the team reviewed work orders and PIPs associated with the system and determined that the problems associated with these had been appropriately addressed by the licensee. Industry operating experience was also reviewed and accessible portions of the system were walked down. No problems were identified based on this review.

b.2 Feedwater Isolation Valves

There were four feedwater isolation valves per unit identified as 1(2)CF-26, 28, 30, and 35. These valves had one feedwater system function identified as CF.6, "Provides two trains of feedwater isolation for each S/G by closing CF valves." This function was not classified as risk-significant by the expert panel. The feedwater isolation valves were unique since each valve has a self-contained, electro-hydraulic subsystem for the hydraulic actuator. For each valve actuator subsystem there were two trains (A & B) of solenoid pilot valves. The problems associated with these solenoid pilot valves and the hydraulic fluid was the cause for the (a)(1) classification. In addition, for each valve, there was a nitrogen charged backup accumulator for closing during emergency service.

The failure of performance criteria of "No Repetitive MPFF" caused the feedwater isolation valves to be classified as (a)(1). A white residue was found in the hydraulic fluid that clogged the pilot valves. The pilot valves were always energized when the feedwater isolation valve was in the open position. Excessive heat from the energized pilot valve caused the hydraulic fluid to degrade at a fast rate. In addition, hydraulic fluid leaked in several valves. These problems were identified in PIPs 1-M97-0259, 1-M97-0747, 1-M97-3409, and 2-M97-1041. PIP 2-M97-1041 contained the corrective action for all valves. A minor modification was implemented to reduce the power (heat) output for each solenoid. The reduction in heat reduced the degradation of the fluid. The team concluded this modification was adequate.

The goal for returning the feedwater isolation valves to (a)(2) status was no failures (MPFF) for a rolling 24 month period. The team concluded the goal was appropriate and the licensee implemented corrective action to address the problem. Review of the raw data for the system (WOs and PIPs) determined that data were being properly captured by the licensee. In addition, a review of industry operating experience determined that it were being appropriately used. No Maintenance Rule problems were identified concerning these valves.

b.3 Reactor Coolant System

McGuire Unit 2 has experienced repetitive maintenance preventible functional failures (MPFFs) of the 2B and 2D reactor coolant pumps (RCPs) within the past year due to degradation of motor stator insulation which caused motor shorts. For this reason, RCPs were classified as (a)(1) by the licensee under the Rule. A replacement and refurbishment schedule for both units' RCPs motors had been established. The licensee has changed insulator testing methods, proposed corrective action to install an on-line monitoring device which will monitor stator electrical capacitance discharge rates to detect voids in motor stator insulation, and established goals to improve RCP performance. In addition, the licensee was evaluating the RCP startup process. Stator insulation breakdown does occur with each motor start and reducing the number of starts coming out of an outage could improve motor reliability. The licensee was also considering an RCS vacuum-fill process that would only require one RCP start in lieu of the three or more starts normally performed to remove air from the RCS. Operating experience from Catawba and South Texas Project was being examined for the vacuum-fill process. The team concluded that corrective action, goals and monitoring to improve RCP motor performance were satisfactory.

In addition, the team reviewed the McGuire Maintenance Rule summary sheets on Maintenance Rule requirements for the RCS. The team found that the licensee had established several different performance criteria for the RCS. In addition, the team noted that the system engineer monitors several key parameters on the RCS to verify system reliability at the component level. This included RCP performance monitoring of motor upper and lower bearing temperatures, seal outlet temperatures, motor stator winding temperatures, motor cooling, motor oil levels and others. The team found that the licensee implemented all requirements of the Maintenance Rule for the RCS.

c. Conclusions

The licensee had considered safety in establishing goals and monitoring for systems and components in an (a)(1) status. Corrective actions for identified problems were appropriate. In general, operating data on the systems were being properly captured in the Maintenance Rule data base, and industry-wide operating experience was considered, where appropriate.

M1.7 Preventative Maintenance and Trending for (a)(2) SSCs

a. Inspection Scope (62706)

Paragraph (a)(2) of the Rule states that monitoring as required in paragraph (a)(1) is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through the performance of appropriate preventative maintenance, such that the SSC remains capable of performing its intended function.

The team reviewed selected SSCs listed below for which the licensee had established performance criteria and was trending performance to verify that appropriate preventative maintenance was being performed, such that the SSCs remain capable of performing their intended function. The team evaluated the use of industry-wide operating experience, trending of SSCs against performance criteria, and corrective action taken when SSCs failed to meet performance criteria, or when an SSC experienced an MPFF.

The team reviewed program documents and records for selected SSCs the licensee had placed in the (a)(2) category in order to evaluate this area. The team also discussed the program with the Maintenance Rule coordinator, system engineers, maintenance supervisors, and other licensee personnel.

b. Observations and Findings

b.1 Structures

The licensee's program for inspection of structures under the Maintenance Rule was documented in Procedure EDM-410, "Inspection Program for Civil Engineering Structures and Components", Revision 1. Review of the program and discussions with the responsible engineers determined that the program was established in accordance with the guidance provided in NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2, and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2. The licensee had established performance criteria for monitoring structures under (a)(2) of the Rule and had provided criteria for moving structures from (a)(2) to (a)(1) of the Rule. The licensee's criteria for structures was consistent with the current guidance provided by Regulatory Guide 1.160, Revision 2. The team also determined that the licensee had completed the historical review of structures in accordance with NUMARC 93-01.

Condition monitoring of applicable structures was performed in accordance with Site Plan MG-97-0399, which has a final completion date of March 1, 1998. At the time of the inspection the licensee was approximately 50% complete with the baseline inspections. The team reviewed the inspection findings for the completed inspections and found the documentation to be excellent. Structures will be inspected on a five year frequency as required by EDM-410. Walkdown inspections and review of work orders by the team confirmed acceptable documentation of structures examined by the licensee.

The team concluded that the licensee's structures program established under the Rule was comprehensive and effectively implemented. This program was considered a strength by the team.

b.2 Generator Stator Cooling

The licensee classified the generator stator cooling system as a non-risk-significant Maintenance Rule system. The function of this closed loop system was to maintain quality water for cooling and monitoring the generator stator and related components, such that a reactor trip, safety system actuation or 10% plant transient does not occur.

The system was monitored against plant level performance criteria. Review of historical data and work orders revealed that appropriate corrective actions had been taken for the minor problems experienced by this system. Walkdown inspections revealed that the system was being properly maintained. No deficiencies were noted concerning this system.

b.3 4.16KV Essential Auxiliary Power

This system consisted of two trains of safety-related 4.16KV power. It included the load side breakers from the 6.9KV buses and the 6.9/4.16 transformers to the line side of the 4.16KV buses. It did not include any of the load side breakers. These load side 4.16kV breakers were included in the system receiving the 4.16KV power. The system was classified as (a)(2) for both units since performance criteria had been met.

The team reviewed the performance criteria and performance monitoring for this system and determined that the criteria were established in accordance with the NUMARC 93-01 guidance. The licensee was monitoring both reliability and unavailability for the system.

The team reviewed the system's Maintenance Rule summary sheet, corrective work orders, preventive maintenance work orders, and PIPs to verify the system was properly classified. The team concluded the system was well maintained and was appropriately classified as (a)(2).

Inspection determined that there was no system status report in the "Engineering Support Program" for this system. Discussion of this issue with the licensee determined that status reports were not available for many of the Maintenance Rule systems. This led to the identification of a weakness concerning trending of system data, in that, no uniform method existed to track and report system status, and status reports did not exist for all Maintenance Rule systems.

b.4 The 6.9KV Normal Auxiliary Power

This system was not classified as safety-related or risk-significant. The system consisted of two trains that included the line side breakers from the 24KV/6.9KV transformers ATA and ATB. The main function of the system was to provide power from the 6.9KV/4.16KV transformers to the 4.16KV system. The Maintenance Rule functions were: (1) EPB.1, "Maintain Power to 4160V Essential Auxiliary Power System," (2) EPB.2, "Maintain Reactor Colant Safety Breakers," (3) EPB.3, "Maintain Power to Large Unit and Station Loads," (4) EPB.4, "Provide Means to Start and Trip RCPs," and (5) EPB.6, Maintain Power to Non-essential Systems". Function EPB.5 was deleted. General plant level performance criteria were applied to the system.

The team reviewed the system's Maintenance Rule summary sheet, corrective work orders, preventive maintenance work orders, and PIPs to verify the system was properly classified. The team concluded that the system was well maintained and was appropriately classified as (a)(2). As was the case with the 4.16KV system, there was no status report in the "Engineering Support Program" for the system. See Section M.1.7.b.3 for a further discussion of this weakness.

b.5 Auxiliary Feedwater System

The auxiliary feedwater system (AFW) system was being monitored under (a)(2) of the Maintenance Rule with performance and condition monitoring criteria established. The actual unavailability data for AFW pump trains was much lower than the unavailability performance criteria limit of 4% (i.e., 525 hours per cycle) for the AFW system. The team also found that this unavailability performance criteria was larger than the assumptions used in the PRA (i.e., 1.4% for the turbine driven AFW pump train and 1.2% for the motor driven AFW train). The licensee performed a sensitivity study to evaluate the adequacy of these criteria. Based on this study, the unavailability criteria may be lowered by the licensee (see Section M1.2 of this report for additional details).

The team found that licensee PIPs and work orders (WO) over the past two years showed that no Maintenance Rule functional failures or MPFFs had occurred on the AFW system. With the one Maintenance Rule scoping exception previously noted (see Section M1.1 of this report), the team found that the licensee was implementing all Maintenance Rule requirements for the AFW system.

b.6 Ice Condenser System

The team found that the Unit 2 ice condenser system experienced lower inlet door problems which did not meet TS 3/4.6.5.3.1. Ten of 48 doors did not meet surveillance test requirements performed on the ice condenser system on July 17, 1997. The team reviewed PIP 2-M97-2686 and found that 10 lower inlet doors were declared inoperable due to uplift of the concrete floor. The concrete heave caused interference between the door frame metal flashing and the lower inlet doors. The licensee attributed the problem to several ice melt events that resulted in water being absorbed by the concrete. The licensee was also aware of industry experience concerning the occurrence of a similar problem at the Sequoyah plant in 1992. Also, recent degraded performance of the Unit 2 floor cooling system had aggravated this condition. A special NRC inspection was performed for this issue, and the results were documented in Inspection Report 50-370/97-16.

Immediate corrective actions included a design modification which removed a portion of the door frame flashing. This left a minimum clearance of at least 2.25 inches in every bay between the floor and the remaining flashing. The licensee has also measured the floor to monitor future floor warping.

The team reviewed the SSC Maintenance Rule summary sheets for the ice condenser system and noted that there was no function for the floor cooling system. Based on site-specific and industry operating experience, the licensee's expert panel took corrective action to add the ice condenser floor cooling system to the scope of the Maintenance Rule. In addition to TS surveillance testing, the licensee established condition monitoring criteria for the floor. The licensee was also analyzing proposed design changes which would change the operating characteristics of the floor cooling system. The licensee was still evaluating if this was a Maintenance Rule functional failure or an MPFF of the ice condenser system and if the ice condenser floor cooling function should be moved to the (a)(1) category. The team concluded that the licensee was taking corrective actions to improve lower inlet door performance; however, long-term corrective actions had not been developed.

b.7 Hydrogen Mitigation System

The team reviewed the Maintenance Rule summary sheets for the hydrogen mitigation system (HMS) and found that the licensee was monitoring both trains of the HMS along with individual hydrogen igniter glow plugs for this system under the scope of the Maintenance Rule. If two or more hydrogen igniters in a train were to fail, the licensee would declare the HMS train inoperable in accordance with TS. The team reviewed hydrogen igniter failures and train unavailability data tracked by the system engineer for the HMS, PIPs forms 1-M97-2922 and 1-M96-1864, and voltage burn-in testing failure rates for commercial grade dedicated hydrogen igniters. The team noted that the Unit 1 HMS had experienced one hydrogen igniter failure in each train in the last year. The licensee established performance criteria of less than 3 MPFFs and 4% unavailability (i.e., approximately 525 hours) per train per cycle. Based on this data, the team concluded that the licensee was appropriately monitoring and maintaining the HMS under (a)(2) of the Maintenance Rule.

c. Conclusions

For (a)(2) systems, the team concluded that, in general, performance criteria were properly established; industry-wide operating experience was being used, where appropriate; corrective action was taken when SSCs experienced a functional failure; and operating data were being properly captured. The structures program established under the Rule was assessed as a strength, due to the fact that it was comprehensive, and was being effectively implemented. A weakness was identified concerning trending of system data, in that, no uniform method existed to track and report system status, and that status reports did not exist for all Maintenance Rule systems.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Material Condition Walkdowns

a. Inspection Scope (62706)

During the course of the reviews, the team performed walkdowns of the following systems and plant areas, and observed the material condition of the listed SSCs.

- Auxiliary Feedwater System
- 6.9KV Normal Auxiliary Power
- 4.16KV Essential Auxiliary Power
- Feedwater Isolation Valves

- Refueling Water System outside containment
- Diesel Generator Buildings
- Safe Shutdown Facilities
- Spent Fuel Buildings
- Generator Stator Cooling System

b. Observations and Findings

The team performed material walkdowns on selected portions of each of the SSCs that related to the areas inspected. Housekeeping in structures and in the general area around the systems and components was acceptable. Piping and components were painted, and very few indications of corrosion, oil leaks, or water leaks were evident. All minor concerns identified by the team were immediately addressed by the licensee.

c. Conclusions

In general, walkdown of structures and systems determined that they were being satisfactorily maintained.

M7 Quality Assurance in Maintenance Activities

M7.1 Licensee Self-Assessment

a. Inspection Scope (62706)

The inspection team reviewed the following self-assessments to verify that Maintenance Rule independent assessments were conducted and the findings were addressed:

- NEI assisted assessment of all three Duke Power nuclear sites (March 1995).

- Quality assurance assessment of the Maintenance Rule implementation (Assessment Report No. SA-96-66) dated June 26, 1996.

- McGuire Nuclear Station Maintenance Rule Site Assessment SA-97-36 (July 1997).

The licensee's corrective action to several of the July 1997 self-assessment findings were still in process at the time of this inspection.

b. Observations and Findings

In general, the quality of the audits and assessments were good. They were detailed and addressed Maintenance Rule requirements and related items. The reports identified a number of findings and made recommendations for improvement of the program. Corrective actions were appropriately implemented. The team considered the last assessment to be the most thorough, and its findings and recommendations were appropriate and provided the proper guidance for strengthening a good program.

c. Conclusions

Audits and self-assessments of the Maintenance Rule were considered to be a program strength.

III. ENGINEERING

E2 Engineering Support of Facilities and Equipment

E2.1 Review of Updated Final Safety Analysis Report (UFSAR) Commitments (62706)

A recent discovery of a licensee operating their facility in a manner contrary to the UFSAR description highlighted the need for a special, focused review that compared plant practices, procedures and parameters to the UFSAR descriptions. While performing the inspections discussed in this report, the team reviewed the applicable portions of the UFSAR that related to the areas inspected. The team verified that the UFSAR wording was consistent with the observed plant practices, procedures and parameters.

E4 Engineering Staff Knowledge and Performance

E4.1 Engineer Knowledge of the Maintenance Rule

a. Inspection Scope (62706)

The team interviewed licensee system owners (system engineers) for the SSCs reviewed in paragraphs M1.6 and M1.7 to assess their understanding of the Maintenance Rule and associated responsibilities.

b. Observations and Findings

System engineers' knowledge of the assigned systems was excellent, and their knowledge of the Maintenance Rule was adequate. However, the design of the program for implementation of the Maintenance Rule was not heavily dependent on system engineers for implementation. Therefore, when system engineers were questioned concerning their responsibilities as delineated in Procedure EDM-210, weaknesses were encountered.

c. Conclusions

System engineers' knowledge of their systems was excellent and their knowledge of the Maintenance Rule was adequate. Knowledge of their responsibilities under Duke Power Procedure EDM-210 was not strong.

V. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

The team leader discussed the progress of the inspection with licensee representatives on a daily basis and presented the results to members of licensee management at the conclusion of the inspection on August 22, 1997. The licensee acknowledged the findings presented.

PARTIAL LIST OF PERSONS CONTACTED

LICENSEE:

P. Abraham, Manager, PRA NGO
B. Barron, Site Vice President
J. Boyle, Electrical/Civil Engineering Manager
G. Barker, Work Window Manager
D. Brewer, Senior Engineer (PRA) NGO
P. Herran, Engineering Manager
P. Kowalewski, Engineer NGO
T. Pedersen, Maintenance Rule Coordinator
J. Pressley, Operations Matrix Support, Expert Panel Member
K. Thomas, Work Control Superintendent
B. Travis, Mechanical Systems Engineering Manager

NRC:

J. Coley, Reactor Inspector, Maintenance Branch, RII
M. Franovich, Resident Inspector, McGuire, RII
P. Fredrickson, Maintenance Branch Chief, RII
R. Gibbs, Senior Reactor Inspector, RII
M. Miller, Reactor Inspector, Engineering Branch, RII
M. Sykes, Acting Senior Resident, McGuire, RII
F. Talbot, Operations Engineer, NRR
S. Wong, NRC Contractor

LIST OF INSPECTION PROCEDURES USED

IP 62706	Maintenance Rule

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
CDP	Core Damage Probability
EDM	Engineering Directives Manual
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
HMS	Hydrogen Mitigation System
LCV	Lower Containment Ventilation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
MPFF	Maintenance Preventable Functional Failure
NEI	Nuclear Electric Institute
NGO	Nuclear General Office
NRC	Nuclear Regulatory Commission
NSD	Nuclear System Directive
PIP	Problem Investigation Process

PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
SI	Safety Injection
SRO	Senior Reactor Operator
SSC	Structures Systems and Components
STA	Shift Technical Advisor
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
WO	Work Order
WPM	Work Process Manual

LIST OF PROCEDURES REVIEWED

McGuire Nuclear Station Unit 1 Individual Plant Examination (IPE), November 4, 1991.

Nuclear System Directive NSD-310, "Requirements for the Maintenance Rule," Revision 1.

Engineering Directives Manual EDM-210, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants or the Maintenance Rule," Revision 5.

Engineering Directives Manual EDM-410, "Inspection Program for Civil Engineering Structures and Components", Revision 1

Work Process Manual WPM-601, "Innage Management", Revision 4

Work Process Manual WPM-607, "Maintenance Rule Assessment of Equipment Removed from Service," Revision 3.

McGuire Site Directive MSD-403, "Shutdown Risk Management Guidelines," Revision 7.

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NRC Maintenance Rule Baseline Inspection McGuire Nuclear Site August 18-22, 1997

ENCLOSURE 2

Pete Herran

Engineering Manager

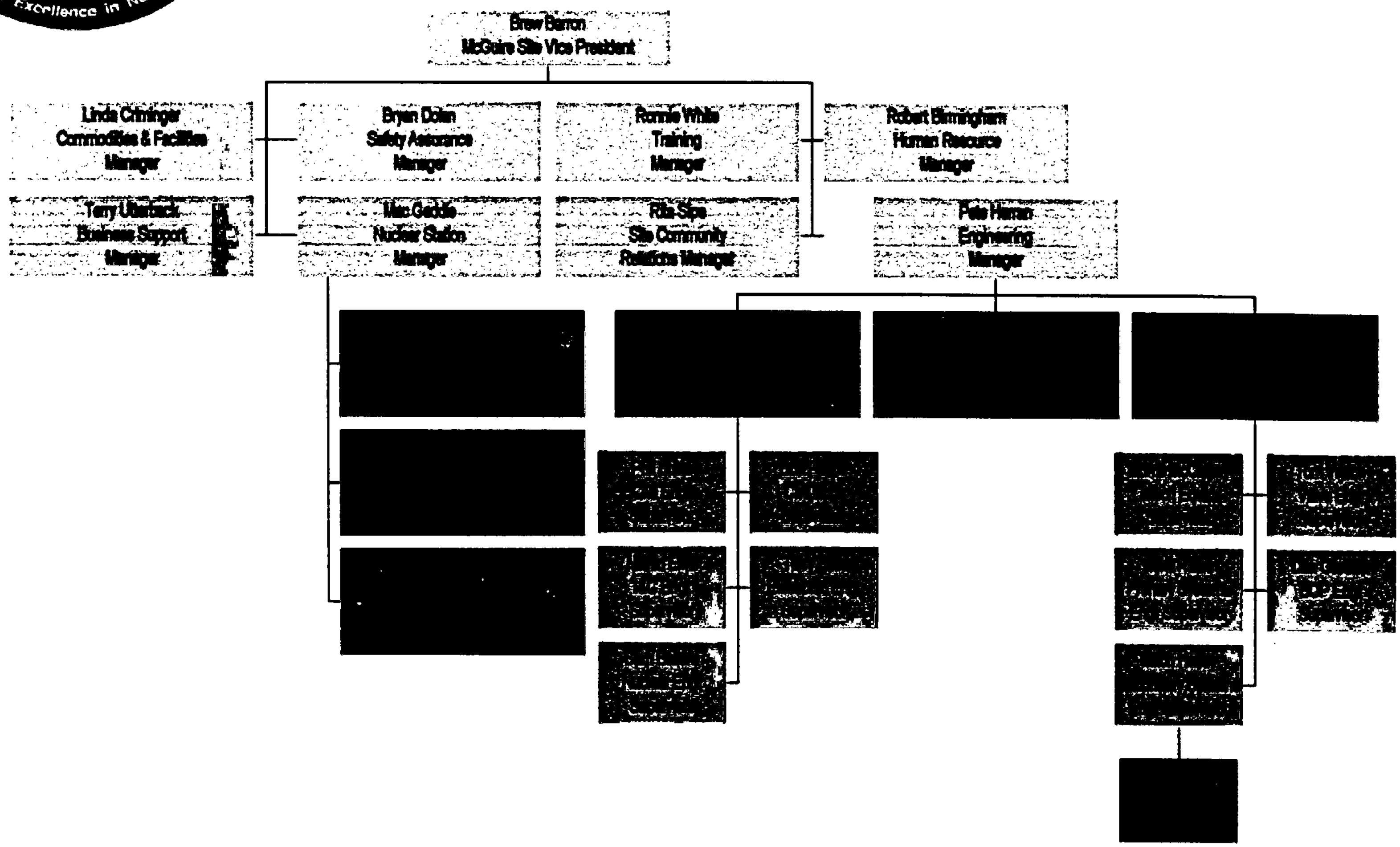
McGUIRE NUCLEAR STATION
MAINTENANCE RULE
AUGUST 18, 1997
ENTRANCE PRESENTATION AGENDA

Introduction	Pete Herran
Maintenance Rule Implementation History	Pete Herran
Uniqueness of McGuire	Pete Herran
Maintenance Rule Program	Terry Pedersen
PRA Interface	Duncan Brewer

ENTRANCE PRESENTATION AGENDA (Continued)

A(1) SSC History	Terry Pedersen
A(3) Portion of the Rule	Terry Pedersen
Self Assessments and Results	Terry Pedersen
Site Management Involvement	Pete Herran
Summary	Pete Herran

SITE ORGANIZATION STRUCTURE



MAINTENANCE RULE IMPLEMENTATION HISTORY

- Two Phase Implementation
  - Maintenance Rule Project Team
  - Maintenance Rule Working Group
- Project Team Purpose
  - To Develop a Maintenance Rule program that complies with 10 CFR 50.65 and Reg Guide 1.160 and NUMARC 93-01 the industry guidance document
- Working Group Purpose
  - To ensure consistent program implementation between the Duke sites.

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IMPLEMENTATION PROGRAM

MAINTENANCE RULE WORKING GROUP (MRWG)

Bruce Travis
Mechanical Systems
Engineering BEST
Sponsor

Mike Carwile
Secondary Systems
Engineering BEST
Sponsor

Pam Kowalewski
MRWG Leader
NGO Mechanical Systems

Robert Knoerr
Maintenance Rule Coordinator
Oconee

Brian Felker
Maintenance Rule Coordinator
Catawba

Terry Pedersen
Maintenance Rule Coordinator
McGuire

Duncan Brewer
NGO Severe Accident Analysis
PRA Expert

Charles Sawyer
NGO Work Control and
Human Performance

MAINTENANCE RULE REQUIREMENTS INCORPORATED INTO EXISTING PROGRAMS

- Problem Investigation Process (PIP)
- Tech Spec Action Item Log (Unavailability)
- Work Management Program (Risk Management)

UNIQUENESS OF McGUIRE

- Two Unit Westinghouse 4 Loop Ice Condenser Plant (3411 MW per Unit)

- New Unit 1 Steam Generators

- Corrective Action Program (Problem Investigation Process, PIP)

Terry Pedersen

Site Maintenance Rule Coordinator

MAINTENANCE RULE PROGRAM

- Administrative Procedures
- Maintenance Rule Process
- SSC Breakdown
- Performance Criteria

ADMINISTRATIVE PROCEDURES

[redacted]

[redacted] [redacted] [redacted] [redacted]

ADMINISTRATIVE PROCEDURES

- Written to NUMARC 93-01, Revision 2 Requirements

- Exceptions to NUMARC 93-01
  - Number of Reactor Trips per Cycle replaces Unplanned Reactor Trips per 7000 Hours
  - Forced Outage Rate and Plant Transients Load Changes >10% MWe Full Electric Load replaces Unplanned Capability Loss Factor

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MAINTENANCE RULE RESPONSIBILITIES

<u>SSC Engineers</u>
Monitoring SSC Performance and Perform
A1 SSC Evaluations and Corrective Actions

<u>Work Control</u>
PRA Risk Matrix for Scheduling
Equipment Out of Service

<u>Operations</u>
PRA Risk Matrix for
Authorization of Maintenance

<u>Safety Review Group</u>
PIP Screening for Maintenance
Rule Functional Failures

<u>Operating Experience Assessment Group</u>
Evaluation of Industry Events

<u>Maintenance Rule Coordinator</u>
Program Coordination and
A(1) Determinations

MAINTENANCE RULE
SSC BREAKDOWN

Total Systems	247
Systems in the Maintenance Rule	169
Risk Significant Systems	48
Total Structures	67
Structures in the Maintenance Rule	36
Risk Significant Structures	3
Total Maintenance Rule [illegible]	205

MAINTENANCE RULE SSCs AND PERFORMANCE CRITERIA

- Expert Panel
  - Review SSC Scoping
  - Make Risk Significance Determinations
  - Approve Performance Criteria
  - Approve the PRA Risk Matrix and Changes
- Performance Criteria
  - Plant Level Performance Criteria
  - System Level Performance Criteria
  - Condition Monitoring for Structures

PLANT LEVEL PERFORMANCE CRITERIA

Repetitive MPFFs	0 Repetitive MPFFs	All Maintenance Rule SSCs
Reactor Trips	<2 Trips	All Maintenance Rule SSCs
Safety System Actuations	<2 SSAs	All Maintenance Rule SSCs
Loss of DHR Events	0 Events	All Maintenance Rule SSCs
Plant Transients	< 4 Events of Unplanned Load Changes >10% MWe	All Maintenance Rule SSCs
Forced Outage Rate	8%	All Maintenance Rule SSCs

* Repetitive MPFFs Can Cross Unit Boundaries Over a Period of 24 Months

SYSTEM LEVEL PERFORMANCE CRITERIA

Risk Significance Functions: Requiring **100%** Availability and Reliability	No MPFFs per Maintenance Rule System	Risk Significance Functions requires 100% Availability
Risk Significance Functions: Requiring **R/S VERY HIGH** Availability and Reliability	No MPFFs per Maintenance Rule System	Risk Significance Functions requires 99.8% Availability
Risk Significance Functions: Requiring **R/S HIGH** Availability and Reliability	< 2 MPFFs per Maintenance Rule System	Risk Significance Functions requires 98.0% Availability
Risk Significance Functions: Requiring **R/S MEDIUM** Availability and Reliability	< 3 MPFFs per Maintenance Rule System	Risk Significance Functions requires 96.0% Availability
Risk Significance Functions: Requiring **R/S LOW** Availability and Reliability	< 3 MPFFs per Maintenance Rule System	Risk Significance Functions requires 94.0% Availability
Risk and Non-Risk Significance Grouped Functions (Both)	< 5 MPFFs per any Maintenance Rule System Grouping	Availability is not required for Non-Risk Significant Maintenance Rule SSCs

PERFORMANCE CRITERIA CONDITION MONITORING FOR STRUCTURES

Acceptable	Structure is A(2)	All Maintenance Rule Structures
Unacceptable	Structure is A(1)	All Maintenance Rule Structures

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Duncan Brewer

Nuclear General Office,
Severe Accident Analysis

McGUIRE's PRA

- Level 3 PRA with Internal and External Events
- Small Event Tree, Large Fault Tree Methodology
- PRA Limitations
- Important Systems FWST, RN and 4.16 kV Essential Power

PRA CALCULATIONS USED FOR RISK SIGNIFICANT SSCs

- Risk Achievement Worth (RAW)
- Risk Reduction Worth (RRW)
- Core Damage Cut Sets

PRA AND PERFORMANCE CRITERIA

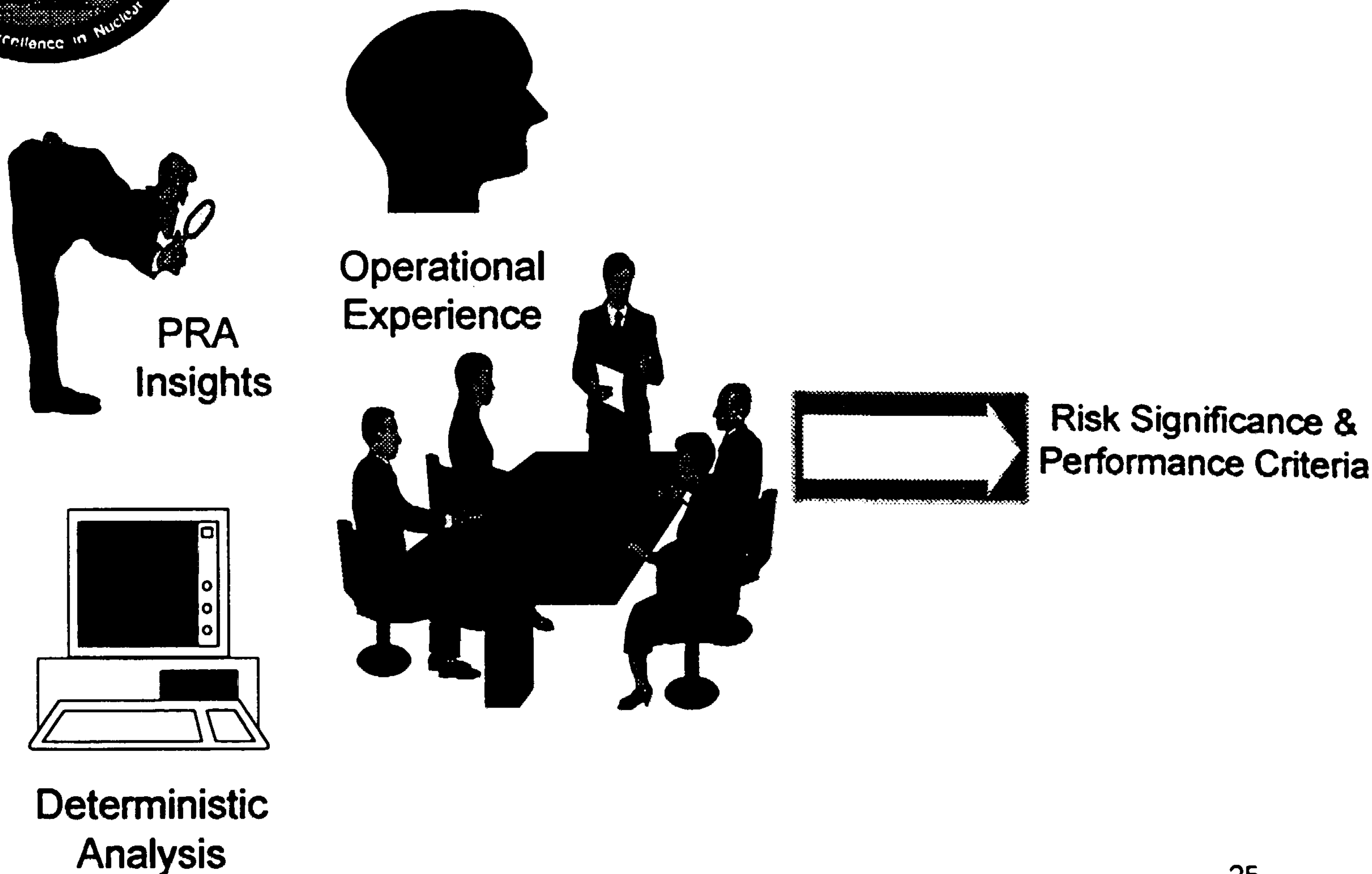
- PRA was Used to Determine Unavailability Performance Criteria

- PRA Insights were Used to Develop Reliability Performance Criteria for Risk Significant Systems

PRA MATRIX

- A Matrix is Used to Assess Risk of Taking Equipment Out of Service
- PRA Insights Were Included in Matrix Development
- PRA Calculations for SENTINEL Development Will be Used to Validate the Matrix

INPUTS FOR EXPERT PANEL DECISIONS



FEEDBACK PROCESS

Living
PRA

Maintenance
Rule
Data

Terry Pedersen

Site Maintenance Rule Coordinator

ON-GOING MAINTENANCE RULE ACTIVITIES

- Current A(1) List
- PRA Risk Matrix
- Periodic Assessments
- Self Assessments

A(1) LIST
JULY 31, 1997

AD	Standby Shutdown (Diesel) System	0	Adverse Trend	Adverse trend on transfer pump
EMF	Radiation Monitoring	1&2	Change in Performance Criteria	Engineering Review
CF	CF Isolation Valves	1&2	Total MPFFs	Multiple failures and load reductions due to CF isolation valves
CB	Auxiliary Boiler Feedwater	1	Safety System Actuation	Failure of boiler feed pump
CF	Feedwater	1	Safety System Actuation	Doghouse actuation
SM	Main Steam	1	Repetitive MPFFs	Steam leakage on high pressure turbine
FW	Refueling Water	2	Unavailability	Problems with heat trace
NC	Reactor Coolant	2	Forced Outage Rate, Reactor Trips & Repetitive MPFFs	NC Pump Motor 2B shorted to ground

WPM 607 Attach. 607.6.5 MNS PRA Matrix (Rev.3, 02/03/97)			**McGuire Unit #**					**Risk Assessment Matrix**								Prepared By					
CAUTION:	This Matrix Does NOT Replace the Technical Specifications.		Date/Time		From									Unit							
	Tech specs should be reviewed prior to using the matrix.		Electrical				SG Cooling		Cooling Water		Reactor Coolant		ECCS				Containment				SSF
	Matrix applies Modes 1, 2, and 3		SYD	EDG	EPC	SAT	CAmp	CAtp	KC	RN	NCb	NCp	ND	NI	NV	FWST	CNTh	CNTi	ICE	NS	SSF
Electrical	230/525 kV Switchyard Systems (Note1)	SYD																			
	Emergency Diesel Generator System (Note 2)	EDG																			
	4.16 kV Essential Power (Note 2)	EPC																			
	Power to 4160 VAC Standby Transformer From Other Unit	SAT																			
SG Cooling	CA System Motor Driven Pump (Note 2)	CAmp																			
	CA Turbine Driven Pump	CAtp																			
Cooling Water	Component Cooling System (Note 2)	KC																			
	RN System (Note 2)	RN																			
Reactor Coolant	NC Interfacing System Pressure Boundary	NCb																			
	NC Pressure Control (Less than 1 Nitrogen Backed PORV per Train Available)	NCp																			
ECCS	ND System (Note 2)	ND																			
	NI System (Note 2)	NI																			
	NV System (Note 2)	NV																			
	Refueling Water Storage Tank	FWST																			
Containment	Containment Hydrogen Control Functions (Note 2)	CNTh																			
	Containment Isolation Functions	CNTi																			
	Ice Condenser / Divider Barrier Seal (Note 2)	ICE																			
	Containment Spray System (Note 2)	NS																			
SSF	Standby Shutdown Systems (Note 1)	SSF																			

Note 1 May affect the Matrix for both units

Note 2 This system or function may be affected by a support system being inoperable. If the system is still functionally available, DO NOT highlight it on the Matrix. If the system is unable to perform its Maintenance Rule Risk Significant function, then DO highlight it on the matrix. Further guidance is provided in WPM 607 Sections 607.5.5 and 607.5.6

Same System

PRA Not Allowed

PRA Interaction (2 or More Not Allowed in Same Row or Column)

PERIODIC ASSESSMENTS

- Historical Evaluation
  - 3 Year Historical Data Review (January 1, 1993 through December 31, 1995)
  - Divided into two 18 month cycles
  - Four A(1) SSCs on July 10, 1996
- Unit 1 Periodic Assessment (under development)
  - Cycle 11 (January 26, 1996 - May 19, 1997)
  - Proposal to conduct a site Periodic Assessment versus a Unit Periodic Assessment

SELF ASSESSMENTS

- NEI Assist Visit (March 1995)
- Site Assessment (May 1996)
- Site Assessment (July 1997)

- Strengths Identified:
  - PRA Matrix Development and Ownership of the Program
  - Ready Retention of SSC Data From Maintenance Rule Database

- Significant Challenge for Improvement:
  - Additional Training for Site Personnel

Pete Herran

Engineering Manager

MANAGEMENT INVOLVEMENT

- Daily PRA Matrix Status is Discussed in Operations Shift Managers Meeting and Management Focus Meeting
- A(1) List is Part of Monthly Plant Systems Health Report That is Discussed in the Management Focus Meeting
- Operational Excellence Quality Steering Team Monthly Reviews the Maintenance Rule Portion of the Monthly Systems Health Report
- Monthly A(1) Report Available to the Site on the Internet

FUTURE

- Evolving and Learning Process
  - Self Correcting Rule
  - Future Process Improvements are Expected
    - PRA Updates
    - SENTINEL Risk Assessment Program